UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

July 8, 1991

Docket No. 50-029

Mr. Frederick A. Latendorf
94 Brookside Avenue, J. P.
Boston, Massachusetts 02130

Dear Mr. Latendorf:

I am writing in response to your letter of June 20, 1991, in which you expressed concerns about the Yankee Rowe reactor vessel and license renewal for the Yankee Rowe facility.

The U.S. Nuclear Regulatory Commission (NRC) staff has evaluated the Yankee Rowe vessel issues and has determined that the vessel condition continues to provide adequate protection of the public health and safety. On August 31, 1990, the NRC staff issued a safety assessment of the Yankee Rowe reactor vessel (copy enclosed) and concluded that there was reasonable assurance that the facility could be operated for an additional operating cycle, currently expected to be completed in early 199[illegible]. In determining to authorize operation for the current cycle, the staff thoroughly considered the views of Dr. Pryor N. Randall, an NRC technical staff member who disagreed with the staff's August 31, 1990, safety assessment. Subsequently, the NRC's Advisory Committee on Reactor Safeguards (ACRS) reviewed the Yankee Rowe reactor vessel issues, including the views of Dr. Randall. The ACRS reported favorably regarding operation of the facility for the additional operating cycle.

On June 4, 1991, the Union of Concerned Scientists (UCS) and the New England Coalition on Nuclear Pollution (NECNP) petitioned the NRC, pursuant to the provisions of Section 2.206 of Title 10 of the Code of Federal Regulations (10 CFR 2.206), to immediately shut down Yankee Rowe. By letter of June 25, 1991 (copy enclosed), the Director of the Office of Nuclear Reactor Regulation responded to the Petitioners, stating that the NRC staff had found that the Yankee Rowe reactor vessel does not pose an undue risk to the public health and safety. Accordingly, the Director determined that Petitioners' concerns did not warrant immediate action to shut down Yankee Rowe. Consistent with 10 CFR 2.206, the NRC will further address the specific issues raised by the Petitioners and the staff is preparing the detailed response.

I want to emphasize that the Commission has made no decision regarding operation of the Yankee Rowe facility after the completion of the current operating cycle. The NRC staff will review the substantial technical data to be provided by the licensee and will evaluate the results of inspections to be conducted during the refueling outage before making a decision as to what actions will be necessary before allowing further operation. This decision will be totally independent of any actions regarding license renewal for the Yankee Rowe facility.

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You expressed concerns regarding license renewal for Yankee Rowe. The licensee has not made a final decision nor submitted an application for renewal of its operating license, even though the utility has actively participated in related rulemaking activities. If the licensee does seek license renewal, NRC approval will be dependent upon the sufficiency of the licensee's application, including supporting technical analysis, and an extensive review process that will afford an opportunity for public participation. Yankee Rowe will be required to conform to all applicable requirements throughout the renewal term in accordance with the license renewal regulations in 10 CFR Part 54.

The Commission has scheduled a meeting on July 11, 1991, at its Rockville, Maryland, office to discuss the reactor vessel issues. In addition, the staff will conduct a public meeting with the licensee in the Rowe, Massachusetts, area in the near future. Following the meeting, the public will be afforded the opportunity to address their questions and concerns to the NRC staff.

Sincerely,

Original signed by
Richard H. Wessman, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Safety Assessment dated 8/31/90
2. Ltr to D. Curran fm T. Murley dated 6/25/91

cc: Mr. George Papanic, Jr.
Senior Project Engineer - Licensing
Yankee Atomic Electric Company
580 Main Street
Bolton, MA 01740-1398

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Enclosure 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 31 1990

Docket No. 50-029

Dr. Andrew C. Kadak
President and Chief Operating Officer
Yankee Atomic Electric Company
580 Main Street
Bolton, Massachusetts 01740-1398

Dear Dr. Kadak:

SUBJECT: YANKEE ROWE REACTOR VESSEL

By letter of July 5, 1990. you submitted for staff review the report, "Reactor Pressure Vessel Evaluation Report for Yankee Nuclear Power Station." This report was in response to our letters of May 1, 7, and 15, 1990. Because of our concerns regarding reactor vessel integrity, we requested information that was needed to assess the effect of vessel operating temperatures, beltline material chemical composition and material surveillance test results. These concerns can affect the conclusions of previous NRC reviews of vessel integrity. Those previous reviews considered postulated Low Temperature Over Pressurization (LTOP) events, Pressurized Thermal Shock (PTS) events, and low irradiated Charpy Upper Shelf Energy (USE).

In your July 5, 1990, submittal, you stated that the RTNDT values for reactor vessel plate and weld metal for the years 1990 and 2000, are below the screening criteria of 270°F and only slightly above the screening criteria for the year 2020. Additional information was provided to support your statements in numerous communications which are listed as references in the attached NRC safety assessment report.

The staff, in its review of your submittal, has concluded that there are substantial uncertainties associated with the weld chemistry and the effects of coarse grain plate material on the shift in the RTNDT reference temperature. These uncertainties could result in reference temperatures significantly higher than the screening criteria specified in the regulations. However, staff calculations, recognizing these uncertainties coupled with estimates of the likelihood of the occurrence of PTS events, lead us to conclude that it is acceptable to operate the Yankee Rowe plant until the end of fuel cycle 21 (approximately February 1992).

Although your July 5, 1990 submittal did not consider LTOP events, the staff has evaluated this scenario based upon additional information provided by your staff. We consider that the systems and procedures implemented at Yankee Rowe, along with estimates of vessel conditional probability of failure provide

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sufficient assurance that the probability of an LTOP event leading to brittle vessel failure is sufficiently low to permit continued operation for an additional cycle.

In your July 5, 1990, submittal, you stated that after performing an additional analysis using the ASME Section XI methodology the USE calculated for the Yankee irradiated plate and weld are 35 ft-lb and 40 ft-lb respectively.

Paragraph IV.A.1 of Appendix G, 10 CFR 50 states that reactor vessel beltline materials must have USE throughout the life of the vessel of no less than 50 ft-lb, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of upper shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. According to staff calculations the USE for the Yankee Rowe vessel could be as low as 35.5 ft-lb. Your USE analysis indicates that the reactor vessel with 35 ft-lb Charpy USE has margins of safety against fracture, equivalent to those in Appendix G of the ASME Code as required by the regulations. The staff has reviewed your analysis and considers that it is acceptable to operate the Yankee Rowe reactor vessel until the end of fuel cycle 21 with 35 ft-lb Charpy USE.

During a meeting on August 21, 1990 you agreed to provide us within 60 days of the meeting date, a proposed plan to address the uncertainties noted herein. The enclosed staff safety assessment addresses these uncertainties in detail. For Yankee Rowe to continue to operate beyond the next operating cycle, we stress the need for you to reduce the uncertainties in the various elements affecting reactor vessel integrity. Your plan should include any appropriate procedural changes, technical specification changes, and sampling and physical tests to ascertain the chemical and physical properties of reactor vessel lower plates and welds. The staff concludes the following actions should be included in your plan:

<u>Long Term Actions to be Completed Prior to Cycle 22 Startup</u>

1. Develop inspection methods for the beltline welds and each beltline plate from the clad to 1 inch from the clad/steel interface to determine if the metal contains flaws.

2. Perform tests on typical Yankee Rowe base metal (0.18-0.20% Cu) to determine the effect of irradiation (f = 1-5E19 n/cm^2), austentizing temperature (1650°F-1800°F) and nickel composition (0.18-0.70 percent) on embrittlement at 500°F and 550°F irradiation temperatures.

3. Determine composition of the circumferential weld metal in beltline by removing samples from the weld.

In addition, prior to Cycle 22 startup, you should install surveillance capsules in accelerated irradiation positions. The capsules are to include materials representing the beltline circumferential weld metal and upper and lower plates.

In addition, you also agreed to the following:

1. Fluence calculations prepared by Westinghouse will be provided by October 1, 1990.

2. Results of peer evaluation of Yankee's July 5, 1990, submittal will be provided within three months.

The results of the staff's review of your July 5, 1990, submittal are included in the enclosed safety assessment.

Sincerely,

Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: See attached

Dr. Andrew C. Kadak

cc:
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ENCLOSURE

SAFETY ASSESSMENT OF YANKEE ROWE VESSEL

I. INTRODUCTION

In a letter dated July 5, 1990 from John D. Haseltine, the Yankee Atomic Electric Company (the licensee) submitted for staff review a report entitled, "Reactor Pressure Vessel Evaluation Report for Yankee Nuclear Power Station." The report was in response to NRC letters dated May 1, 7, and 15, 1990. The staff letters requested additional information, which was needed to assess the effect of vessel operating temperatures, beltline material chemical composition, and material surveillance test results on the integrity of the Yankee Rowe reactor vessel. These concerns have potential impact on prior NRC reviews of vessel integrity resulting from low irradiated Charpy Upper Shelf Energy (USE) and vessel integrity during postulated Pressurized Thermal Shock (PTS) and Low Temperature Overpressurization Events (LTOP) events.

The licensee's justification for operation of Yankee Rowe is that there is adequate assurance that risk of vessel brittle failure is very low. This conclusion depends upon two factors: (1) the frequency of challenges to the vessel, and (2) the probability of vessel failure given a challenge event (conditional vessel failure probability). Brittle failure challenge events fall into 2 general categories: (1) pressurized thermal shock (PTS) events, and (2) low temperature overpressurization (LTOP) events. For both categories the licensee has estimated a very low probability that a vessel failure will occur. The frequency of challenge and probability of vessel failure for PTS and LTOP events are discussed in Section II and Section III respectively. Additional information to support the licensee's conclusion was submitted in References 12 through 23.

II. PRESSURIZED THERMAL SHOCK (PTS) EVALUATION

II.1 Systems Evaluation of PTS Limiting Events

For PTS events the licensee has indicated that risk contributors can be divided into 3 groups: (1) steam line breaks, (2) small break LOCAs, and (3) transients. For each PTS group the event resulting in the most limiting temperature and pressure conditions (from a vessel failure perspective) is considered to be representative for the group. The frequency for a group is the sum of the frequencies for each event in the group.

For PTS the staff's review focused on the following considerations: (1) completeness of the events considered; (2) the adequacy of the thermal hydraulic analyses; (3) adequacy of the event frequency estimates including human error contributions; and (4) adequacy of the limiting events selected.

II.1.1 Completeness of PTS Events Considered

In its PRA submittal on PTS for Yankee Rowe, the licensee performed a systematic evaluation of initiating events (IEs) that could lead to primary system overcooling coupled with primary system repressurization. These IEs were grouped into four major categories. Category I is main coolant system (MCS) induced events. This category of events includes MCS - initiated cool-down events, depressurization events, and injection events, with both the MCS intact and faulted. Category II is secondary system induced events. This category includes events initiated due to steam removal, feedwater flow, steam generator blowdown, and steam/feedwater flow control abnormalities. Category III is general transients which do not directly result in initial MCS cooldown and are not related to support systems but, if followed by other system failures could result in cooldown events. Category IV is events not necessarily resulting in initial MCS cooldown but involving support systems which have the potential to impact other frontline systems which could cause MCS cooldown. The licensee also reviewed the PTS evaluations for H. B. Robinson and Calvert Cliffs performed by Oak Ridge National Laboratories to assure that the Yankee Rowe evaluation took into account sequences found to be significant contributors to thermal shock at these plants. The licensee examined the operating experience at Yankee Rowe (including all the trip logs) and concluded that there has never been an overcooling event at Rowe. The plant design and the Yankee Rowe Probabilistic Safety Study were likewise reviewed to identify any plant unique cooldown sequences.

II.1.2 Thermal-Hydraulic Analyses for Transients Affecting PTS

Based on system and thermal-hydraulic considerations, each of the initiating events were evaluated and the initiators relevant to PTS concerns were identified. Event tree sequences were then developed for each event associated with the relevant initiators concerning PTS. Support systems were treated in a separate auxiliary tree. Quantification of event sequences and endstates was performed based on the system models, dependencies, and human actions. Endstates with frequencies higher than 10^{-8}/reactor year were selected for potential further thermal-hydraulic and fracture mechanics analysis. Based on grouping sequences with similar plant thermal-hydraulic behavior, this process resulted in the final set of initiating events being grouped into three categories with four corresponding event trees: steam line breaks upstream or downstream of non-return valves, small break LOCAs, and transients.

For each of the above identified four event trees, thermal-hydraulic analyses were performed to model the spectrum of overcooling events. The transient downcomer temperature and MCS pressure were calculated and bounding cases affecting PTS concerns were identified.

The licensee used the CEPAC computer code to perform scoping calculations for the events of concern to predict limiting cooldown transients at Yankee Rowe. Based upon pressure and temperature response a small break LOCA of 1 5/16 inches at the reactor coolant pump suction and three cases of main steamline break were found as the limiting transients relative to PTS concerns. These limiting transients were analyzed in greater detail using the RETRAN computer code, the combination of RETRAN and EPRI models, or the combination of RETRAN and REMIX codes. The RETRAN computer code is designed to analyze the response of plant systems during both normal and transient conditions. The licensee's capability of using RETRAN for main steam line break analyses was reviewed and approved by the staff in 1983. The licensee asserted that the CEPAC code is similar but simpler than the RETRAN code. The CEPAC code has not been reviewed by the staff. However, the limiting transients results were not based upon CEPAC calculations. The EPRI model has been used for the non-stagnant flow conditions in the Calvert Cliffs PTS analysis. The REMIX code was used for the SBLOCA case without offsite power available, where flow stagnation occurred. The staff has evaluated the adequacy of the licensee's use of REMIX for the Yankee Rowe plant SBLOCA case. We feel that sufficient conservatism exists in this analysis. There are other conservative assumptions considered in the SBLOCA analysis such as early stagnation in the downcomer area, low decay heat, coincident loss of offsite power and an assumption that all three trains of safety injection are injecting water to the MCS. The first three of these assumptions result in minimal mixing of the cold SI water with the hot primary system water. The fourth assumption maximizes the amount of cold water added to the primary system. The result is a conservative (colder) downcomer water temperature. In the main steam line break cases, there are conservative assumptions applied such as zero power at event initiation, low decay heat, dry steam to the break, coincident loss of offsite power, non-return valve failure, etc. As in the SBLOCA cases these assumptions minimize mixing in the primary system and maximize primary system cooldown. The following design features were found to be significant in the analyses:

- The charging pumps trip on a safety injection signal. This feature helps assure that the maximum repressurization achievable during a LOCA or transient that may initiate safety injection is limited to the shutoff head of the safety injection system (1550 psig).

- The safety injection pumps have relatively low capacity and a shutoff head of 1550 psig when HPSI and LPSI are aligned in series. When not aligned in series the shutoff head is limited to 800 psig.

- There is only one pressurizer PORV. This reduces the probability of a stuck open PORV (relative to two PORVs) initiating a cooldown event.

- There is only one turbine bypass valve, and it has low capacity. This limits the rate of potential cooldown (if the valve fails open).

- The emergency atmospheric steam dump valves have low capacity. This limits the rate of cooldown should the valves fail open.

- The condensate pumps trip following a steam break in the vapor containment.

- Emergency feedwater pumps must be manually started.

- The Emergency Operating Procedures direct the operator, in response to imminent PTS conditions, to stop safety injection pumps and low pressure safety injection pumps if there is sufficient subcooling and pressurizer level.

- Although the plant has primary system loop isolation valves, emergency operating procedures only require their operation during a steam generator tube rupture in order to isolate the faulted generator. For other LOCAs inside the vapor containment, the operators are instructed to not isolate the break location. Isolation of a break could result in significant repressurization.

- The feedwater pumps trip on reactor scram or low suction pressure. Above 15 percent power, operators are instructed to isolate feedwater flow by closure of the feedwater regulating valves and the feedwater motor-operated isolation valves. These measures limit the chance and severity of an overcooling event caused by overfeed of the steam generators.

Yankee Rowe is also unique in the large number of ways in which water can be supplied to the steam generators. Among these multiple paths, all flow sources however, are dwarfed in volume by the boiler feedwater pumps. The feedwater control system has independent controls for each steam generator such that a single failure in the control system would not result in overfeeding more than one steam generator. If another system should begin to supply additional water to the steam generators (e.g., the charging system), the feedwater control system would cut back on the flow from the boiler feedwater pumps to maintain steam generator level. In view of these plant specific features and the modelling assumptions used by the licensee, the staff considers that the thermal-hydraulic analyses are conservative and reasonable. We note that the results are also consistent with other similar analyses such as the Robinson and Calvert Cliffs PTS studies.

II.1.3 Frequency of Cooldown Events Threatening the Vessel

Yankee Atomic has estimated that the frequency of sequences that would significantly challenge the integrity of the reactor vessel due to pressurized thermal shock to be about 5 E-4 per reactor year. Small break LOCAs result in the most limiting thermal hydraulic conditions of any of the sequences analyzed. Yankee Atomic estimated this frequency by partitioning the WASH-1400 small break LOCA frequency (for break sizes between 0.5 and 2 inches) based on the number of pipe segments inside the vapor containment that were between 1 and 2 inches in interior diameter (I.D.). The limiting sequence (combination of frequency and thermal hydraulic conditions) was estimated by Yankee to be a LOCA about 1 5/16 inches I.D. where the estimated minimum downcomer temperature was 151°F and the maximum RCS pressure after cooldown was 670 psi. This analysis did not, however, take into account the possibility of the operator violating his Emergency Operating Procedures and attempting to isolate the break. Such action could lead to an RCS maximum pressure equal to the shutoff head of the safety injection pumps. At the staff's request Yankee Atomic performed an analysis of such a sequence. The licensee concluded that it was not a significant event because of the small amount of small bore piping which is isolable, the frequency of a small break in any location, and the operator training and procedures which direct operators not to isolate breaks inside the vapor containment.

The staff has reviewed the licensee's event frequency estimates in consideration of the plant specific features of Yankee Rowe. The limiting event frequencies are reasonably consistent with val(:s used in other studies. The treatment of human error in the Yankee Rowe PTS PRA is judged to be conservative or non-conservative depending on the timing of the error. The PTS thermal hydraulic analyses indicate that small break LOCAs e worst combination of low primary system temperature, high primary sy re, and high cooldown rate. The staff believes that the licensee's e. te of $5x10^{-4}$ per reactor year as the frequency of a small break LOCA is consistent with the frequency of $1x10^{-3}$ per reactor year typically used in PRAs.

II.1.4 Adequacy of PTS Limiting Events

The licensee performed a systematic review of the Yankee Rowe features in order to identify potential overcooling sequences. The licensee then grouped the possible events on the basis of similarity in thermal hydraulic (TH) response. For each group a limiting event was determined based upon consideration of event frequency and the severity of pressure temperature conditions (relative to vessel failure) resulting from the event. The staff concluded the events considered are reasonably comprehensive, the thermal hydraulic analyses, methods, assumptions and results are reasonable. With regard to the frequency estimates, the most important considerations are the insensitivity to human error and the relative frequency values. The systems failure estimates used are considered to be reasonable because they are consistent with state-of-the-art PRA applications. The event frequencies were also found to be relatively insensitive to human error since the limiting events would not change significantly even if the human error probability (at times greater than 1 hour) changed by a factor of 100. Therefore, based on these systems, thermal hydraulic, and event frequency studies, the staff concludes that there is reasonable assurance that the limiting events have been properly identified.

II.2 PTS Materials Evaluation

II.2.1 Background

The Pressurized Thermal Shock (PTS) rule, 10 CFR 50.61, adopted on July 23, 1985, establishes a screening criterion that is a limiting level of embrittlement beyond which operation cannot continue without further plant-specific evaluation. The screening criterion is given in terms of RT_{NDT}, calculated as a function of the beltline material chemical composition (copper and nickel contents) and the neutron fluence according to the procedure given in the PTS rule, and called RT_{PTS} to distinguish it from other procedures for calculating RT_{NDT}. The greater the amounts of copper, nickel and neutron fluence the higher the RT_{NDT} for the material and the lower its fracture resistance. The screening criterion is 270°F for plates and axial welds and 300°F for the circumferential weld. The rule does not consider the effect of vessel operating temperature and material surveillance test results on the calculated RT_{PTS}. The rule is currently being amended to calculate the RT_{PTS} using the trend curves in Regulatory Guide (RG) 1.99, Rev. 2.

The licensee, in response to our concerns about embrittlement; provided the following significant information:

1. The reported copper and nickel contents of the weld metal are now assumed to be higher, because the actual values are unknown, and the licensee elected to report measurements made for a "sister" vessel, the Belgian BR-3 reactor, instead of previously-reported measurements for a weld in the upper head of the Yankee Rowe vessel.

2. The nominal operating temperature is 500°F, whereas the data base for R.G. 1.99, Rev. 2 and the PTS rule is from reactors that operate at a nominal temperature of 550°F. (Lower irradiation temperature increases RT_{NDT}.)

3. The surveillance data from the Yankee Rowe vessel, all of which date from the late 1960's, show high sensitivity to neutron embrittlement, even

considering the effect of the lower irradiation temperature. These data were known to the AEC but were discounted because the operating temperature in the first few fuel cycles was known to be low (500°F), and there were coast down periods involving low operating temperature of several months duration at the end of the fuel cycles.

II.2.2 Evaluation of Material Properties

The beltline in the Yankee Rowe reactor vessel consists of an upper plate, a lower plate, two axially oriented welds and one circumferentially oriented weld. The only surveillance data from these materials is from the YNPS beltline upper plate. The chemical composition and heat numbers for the upper and lower plates are known. The chemical composition and heat numbers for the axial and circumferential welds are unknown. Eighty-five percent of the accumulated irradiation occurred at a cold leg temperature between 500°F and 520°F. The remaining fifteen percent of the accumulated irradiation occurred at cold leg temperatures less than 500°F.

The staff's estimate and licensee's estimate of the mean value reference temperature in 1990 for each Yankee Rowe beltline material at its peak neutron flux location are tabulated in Table I. The mean value reference temperature is the sum of the unirradiated reference temperature and the increase in reference temperature resulting from neutron irradiation at an irradiation temperature of 500°F. The staff's estimate of the increase in reference temperature was estimated for the peak neutron fluence in 1990 at the inside surface of the reactor vessel. The peak neutron fluence is 2.3×10^{19} n/cm² for the upper shell plate, 2.05×10^{19} n/cm² for the lower shell plate and circumferential welds, and $.38 \times 10^{19}$ n/cm² for the axial welds. The neutron fluences were calculated by the licensee using a methodology documented in letters from G. Papanic, Jr. dated January 22, 1986, October 28, 1986 and February 4, 1987. The staff review of the licensee neutron fluence calculation methodology is documented in a letter to the licensee dated March 10, 1987. The licensee is currently recalculating these fluences. The results of this analysis will not be available before October, 1990.

II.2.2.1 Upper Plate

The licensee's estimate of the increase in reference temperature for the upper plate was derived from Yankee Rowe and BR-3 surveillance data, but did not correct the BR-3 data (irradiation temperature 525-540°F) to account for the lower irradiation temperature (500°F) of the Yankee Rowe reactor vessel. In addition, the licensee doubled the neutron fluence values reported for the Yankee Rowe surveillance data. The licensee did not include the effect of lower irradiation temperature in its analysis because they claim that the coarse grain size of the upper plate surveillance material eliminates the effect of irradiation temperature. The licensee's coarse grain theory is based on an argument that irradiation-induced defects in a coarse grain structure are more stable than irradiation-induced defects in fine grain structures. Since the irradiation-induced defects are more stable in the coarse grain structure, the licensee concludes that the lower irradiation temperature of its reactor vessel will not affect the BR-3 data. Because of very limited surveillance data applicable to the Yankee vessel, the staff does not consider that the licensee has yet substantiated this theory.

A literature survey performed by the staff revealed three reports which indicate irradiation temperature has an effect on neutron irradiation embrittlement. In Reference 1 (Stallman, ORNL), irradiation temperature was found to increase transition temperature by 0.5 to 1.5 degree per degree decrease in irradiation temperature from 550°F, for a heat of A 533-B plate (the 02 plate from the ORNL HSST program). Odette (Ref. 2) has similarly found a factor of 1 degree per degree using a large data base of surveillance data. In addition, Lowe (Ref. 3) has found about 0.7 degree per degree change in irradiation temperature, for Linde 80 welds. Overall, these factors are probably dependent on the composition, processing history, etc. of the steel.

Although, References 1 and 2 do not specifically address coarse grain structures, the staff included the irradiation temperature effect in its evaluation because the licensee has not presented any Charpy data that shows the reference temperature for its plate material does not increase with a decrease in irradiation temperature. The staff estimate of the reference temperature includes a correction for irradiation temperature and is based on the analysis performed by Odette (Ref. 4).

II.2.2.2 Lower Plate

The licensee's estimate of the increase in reference temperature for the lower plate was derived from Yankee Rowe and BR-3 surveillance data, but was not corrected for lower irradiation temperature or the increase in the amount of nickel in the lower plate compared to the amount in the surveillance plate. The lower plate has 0.63 percent nickel and the surveillance plate has 0.18 percent nickel. The licensee believes no correction is necessary because of the postulation that the coarse grain of the plate eliminates the nickel and irradiation temperature effects.

To support the conclusion that the nickel effect may be eliminated for coarse-grain structural material, the licensee reports the conclusions of a Maricchiols (Ref. 5) study. In this study, "Nickel was reported to reduce the damage introduced by neutron irradiation up to a content of about 1.0 percent." This study appears to contradict the results from a statistical analysis of commercial US reactor surveillance data. The results of the statistical analysis of base metal surveillance data is reported Table 2 of RG 1.99, Rev.2, which is contained here as Table 2. This Table indicates that for a particular amount of copper, nickel increases the chemistry factor, which results in an increase in the material's reference temperature (damage), not a decrease as reported in the Maricchiols study. Since the statistical analysis performed to derive the chemistry factor in the tables in RG 1.99, Rev. 2 indicates that there is a nickel effect and the licensee has not provided any data from coarse grain structure material that shows there is no nickel effect, the staff concludes there is a nickel effect.

The staff estimates that an increase in nickel from .18 percent to .63 percent at 500°F irradiation temperature results in an 80°F increase in the reference temperature. This value is based on analysis by Odette (Ref. 4). The staff considers that it is important in order to determine whether longer term operation should be authorized to determine the effect of coarse grain for operating temperature and metal chemistry representative of the Yankee Rowe vessel.

II.2.2.3 Circumferential and Axial Welds

The circumferential weld is one of the critical materials. The axial welds are not because they are exposed to only one-sixth of the peak fluence due to their azimuthal location relative to the core.

The licensee estimated the increase in reference temperature for the circumferential welds using the methodology recommended in RG 1.99, Rev. 2 and a correction factor for irradiation temperature. As discussed previously, the chemical composition of the Yankee Rowe beltline welds is not known. The licensee used the chemical composition of a BR-3 weld to estimate the increase in reference temperature resulting from neutron irradiation. The licensee believes that the amounts of copper (.183 percent) and nickel (.70 percent), reported for the BR-3 weld may be used as estimates for their welds because the BR-3 weld and Yankee Rowe beltline welds were fabricated by the same vendor Babcock Wilcox, using the same process (submerged arc) and the same procedures (copper-plated filler wire with Linde 80 flux). However, this conclusion is not supported by industrial experience. The B&W Owners Group (Ref. 6) evaluated the weld chemistry of Babcock & Wilcox fabricated Linde 80 welds. The reports indicates that the total copper concentration in the weld metal results from a combination of the amount of copper plating and the base filler wire alloy concentration. However, the principle source of copper in the as deposited weld metal is the amount of copper plate. Reference 6 indicates the amount of copper varies from heat of wire to heat of wire. Until the licensee determines the chemical composition of the circumferential and axial welds, the amount of copper in the welds should be considered unknown and bounding values of copper should be used to estimate the effect of neutron irradiation on the weld metal's reference temperature.

The staff used two bases for estimating RT_{NDT} for the circumferential weld. One method uses a set of data compiled by Odette (Ref. 4) for 500°F irradiation, which yields a 370°F value for RT_{NDT}. The other method uses RG 1.99, Rev. 2 methodology, bounding values for copper and nickel, 0.35 percent and 0.70 percent respectively, and 50°F for the irradiation temperature effect. This yields a value of 330°F for RT_{NDT}. Figure 1 (Figure 4 from Reference 1) reports the increase in reference temperature for weld metals and base metals (plates) at irradiation temperature of 500°F. The dashed line has been added to represent the increase in reference temperature for the circumferential weld using the RG 1.99, Rev. 2 bounding method with 50°F correction for the irradiation temperature effect. Since this curve bounds all the existing weld data in the Odette report, this method has been used to estimate values of of reference temperatures for the circumferential and axial weld metal where the amount of copper is unknown and the weld metal is subject to 500°F irradiation temperature.

The predicted value of the reference temperatures in 1990 for the circumferential weld and longitudinal welds are 330°F and 226°F, respectively. These values are for high copper welds. If the chemical analyses of these welds indicates that the amounts of copper are significantly less than 0.35 percent copper and 0.70 percent nickel, the reference temperatures will be significantly reduced. For example, if the circumferential weld had 0.20 percent copper and 0.70 percent nickel, the reference temperature would be 262°F (212°F from RG 1.99, Rev. 2 and 50°F for irradiation temperature effect). Thus, the staff considers that it is important in order to determine whether longer term operation should be authorized to determine the actual chemical composition of the circumferential weld.

II.2.3 Summary

The level of uncertainty is higher for the estimates of RT_{NDT} values for Yankee Rowe than has been encountered for other reactor vessels. Therefore, considering the uncertainty in weld chemistry and the effects of coarse grain, the staff believes the RT_{NDT} for both the lower plate and the circumferential weld should be assumed to be 350°F ± 50°F.

II.2.4 Probabilistic Fracture Mechanics

Although the Yankee Rowe reactor vessel beltline has not received any inservice volumetric inspection, other areas of the reactor vessel have been inspected. These inspections report that the welds do not contain any flaws exceeding the acceptance limits defined by 10 CFR 50.55a and ASME Code Section XI.

In developing the PTS rule, the staff used a "Marshall" distribution (Ref. 7) of flaws. The "Marshall" distribution, which was developed in the mid-seventies, characterized defects in a vessel entering service, including defects considered acceptable according to fabrication codes and undetected during inspection.

The Yankee Rowe reactor vessel beltline was fabriacted using methods and materials similar to other commercially operated reactor vessels except that the clad in the Yankee Rowe reactor vessel is spot-welded and the clad in all other commercially operated reactor vessels is fusion welded. Hence, except for the effect of spot welding, the distribution of flaws in the Yankee Rowe reactor vessel should be similar to the distribution in other commerically operated reactor vessels.

During the Summer 1990 refueling outage, the licensee ultrasonically examined the reactor pressure vessel closure head and upper regions of the pressurizer, which contained spot-welded clad similar to the clad in the reactor vessel beltline. The staff inspector (Ref. 8) concurred with the licensee's evaluation of the ultrasonic data that there was no extension of previously observed cladding cracks into the base metal. This inspection supports the conclusion that postulated cracks in the spot weld in the reactor vessel beltline cladding would not progress into the base metal due to the operation of the reactor vessel and the "Marshall" distribution appears to be applicable for the Yankee Rowe reactor vessel beltline. However, until the licensee performs an inservice inspection of the beltline materials, the conditional failure probability should be increased to account for the uncertainty in service-induced flaws.

To assess the effect of cracks on the probability of failure given the occurrence of a transient event, the licensee utilized probabilistic fracture mechanics analysis. The staff guidance for estimating the conditional probability of reactor vessel failure is provided in Regulatory Guide 1.154.

Thermal and stress analyses for the vessel wall have to be performed. Input for this analysis includes the primary system pressure, the temperature of the coolant in the reactor vessel downcomer, the fluid-film heat transfer coefficient adjacent to the vessel wall, all as a function of time, and the vessel properties. Probability density distribution functions for flaw size, crack initiation fracture toughness, crack arrest fracture toughness, and either the vessel materials nil-ductility reference temperature, or the vessel materials copper and nickel contents, and fast neutron fluence have to be developed. For each transient of interest, many deterministic fracture mechanics analyses have to be performed to determine the number of times the crack penetrates through the vessel wall per 100,000 runs (for example) as a result of the stress level, flaw size, toughness and other variables selected for each run. The calculations are performed with a probabilistic fracture mechanics computer code based on the Monte Carlo simulation technique.

The licensee has performed a probabilistic fracture mechanics analyses for several transients. For example, the licensee performed a sensitivity study that predicts conditional probability of reactor pressure vessel failure is approximately 10^{-3} given the occurrence of a 1.3 inch-diameter small break LOCA event, which they believe is the controlling event, and for the reference temperatures reported in Table 3. The reference temperatures used by the licensee are similar to the values estimated by the staff except for the lower plate. The conditional failure probability for a small break LOCA event for the lower plate with a reference temperature of 325°F is less that 10^{-5}. This plate has a low conditional failure probability at these high reference temperatures because only a small portion of the plate is in the beltline region. Considering the results from the 325°F reference temperature analysis, a mean value of 355°F should not significantly change the conditional failures probability.

When evaluating the results of the licensees sensitivity study one must consider the assumptions used in the analysis. The licensee assumed a "Marshall" distribution of flaws and that cracks would arrest according to the average crack arrest data (Ref. 9). The flaw density distribution function used by the licensee may not be representative of the Yankee Rowe reactor vessel because of its unique spot cladding on the inside surface of the reactor vessel. It also appears that the licensee's analysis may not have adequately accounted for the low upper-shelf energy of the vessel material which affects the "arrest" of initiated cracks. Given these apparent deficiencies and others that have been noted to date, the staff does not accept the licensee's estimate of the conditional failure probability of the reactor pressure vessel. The staff and its contractor are continuing a detailed review of the licensee's analysis. The review of this analysis should be completed by the end of October 1990. The results of this review will be important in determining future action in connection with this license. In view of these uncertainties the staff is unwilling to accept the licensee estimate of conditional vessel failure probability of $1x10^{-3}$ given a specific size small break LOCA. In the meantime the staff judges it would be prudent to assume the conditional probability of reactor pressure vessel's failure to be in the range of 10^{-1} to 10^{-2} .

II.2.5 PTS Conclusions

As discussed above, the staff concludes that there are substantial uncertainties associated with weld chemistry and the effects of coarse grain plate material on the shift in reference temperature. These uncertainties could result in reference temperatures significantly higher than the screening criteria specified in the regulations. Recognizing these uncertainties, the staff concluded that a more conservative range of conditional failure probability (by a factor of 10 to 100 relative to the licensee's estimate) was appropriate. This range when coupled with estimates of likelihood of the occurrence of PTS events and consideration of the plant specific features at Yankee Rowe important to such events, leads the staff to conclude that operation until the end of fuel Cycle 21 is acceptable from PTS considerations. However, additional information to resolve these concerns is needed to determine whether to authorize longer term operation.

III. LOW TEMPERATURE OVERPRESSURIZATION (LTOP)

III.1 Systems Evaluation

In addition to the PTS events described above, another class of transients that could induce fracture in a brittle reactor vessel beltline are low-temperature overpressure (LTOP) events. These events could occur during plant heatup when pumps are being started and there are possibilities for the misalignment of valves and controls following maintenance operations. The occurrence of such events has led to requirements comprising a low setpoint relief valve and control circuitry as described in NUREG/CR-5186, (Ref. 10).

For LTOP considerations analyses are divided into two general categories: (1) mass (water) addition events and (2) energy addition events. In its July 5, 1990 submittal the licensee presented analyses of such events for the Yankee Rowe plant. The analyses were based upon industry wide historical data on LTOP events from 1980-1986 adjusted by consideration of Yankee Rowe specific features. The licensee concluded that the likelihood of vessel challenges from LTOP events was very low.

The staff review in this area emphasized the applicability of historical data to Yankee, impact of Yankee specific LTOP system features; and administrative controls used to minimize human errors.

III.2 LTOP Event Frequency

For LTOP analyses the licensee used the method and data described in NUREG/CR-5186 (Ref. 10). Features important for Yankee relative to the generic data base are:

Feature A: The RHR (Shutdown Cooling System) at YNPS is a dedicated system which is different from most plants. The system is connected to the Main Coolant system through dual isolation valves. The suction to the Shutdown Cooling pump is from the #4 cold leg loop. There are two pumps and heat exchangers for redundancy. There is also a relief valve on both the suction and return lines for overpressure protection.

Feature B: The PORV (in the low setpoint condition) and the shutdown cooling relief valves are required to be operable by Technical Specifications whenever the plant is in the Modes 4 and 5 and the system temperature is less than 300°F. The shutdown cooling relief valves are tested when the plant is operating in Mode 1 and the shutdown cooling system is required to be isolated. The PORV is tested when the plant is in Mode 6 with the reactor head removed.

Feature C: Plant procedures require that power be removed by locking out the breakers for the Main Coolant pumps and the Safety Injection pumps prior to being in a water solid condition. Power is removed from SI pumps below 200°F.

Feature D: The safety relief valves of the shutdown cooling system cannot be automatically isolated once the system is placed into operation because the system isolation valves do not have any automatic isolation capability.

Feature E: During water solid condition operations, a dedicated operator is stationed to prevent or terminate any pressure excursion.

During operation below 300°F, 2 shutdown cooling relief valves and 1 PORV are available to mitigate LTOP events. In this temperature range, and with no credit for human intervention during an event, the licensee estimate of vessel challenge event frequency (events where mitigation systems fail) is $6.5X10^{-5}$ per reactor year. NUREG-5186 reports a frequency of $2.5x10^{-3}$ per reactor year using generic data. The difference is attributable to 2 factors: (1) the availability of an additional relief path at Yankee relative to generic data assumptions; and (2) a power lockout requirement for MCP and SI pumps at Rowe which precludes energy addition events such as were reported in the generic data base.

The staff judges that the specific features of Yankee Rowe would reduce the likelihood of the vessel challenges from LTOP events in the operating range when the PORV is reset to the lower setpoint and the SDC system SRVs are available. An event frequency of 1×10^{-3} per reactor year was therefore chosen as a conservative screening value to assess the importance of LTOP events in this temperature range relative to PTS events.

Between 300°F and 330°F the SDC system is isolated, and above 380°F and 450 psig the PORV is reset to 2500 psig. For all temperatures greater than 180°F a pressurizer bubble is required. In the range of 300°F to 450°F a dedicated operator is required whose only responsibility is LTOP protection (by maintaining a 400 psi margin to the Appendix G curve). Power is also removed from 2 of 3 safety injection pumps at these conditions and all SI pump switches must be in pull to lock. Inadvertent SI (which could cause a maximum pressure of 1550 psig) would therefore require a spurious SI signal plus failure to have the SI pumps in pull to lock. In addition, the auto safety injection signal is blocked until 1800 psig. The licensee concluded that the most probable LTOP challenge in this range (T greater than 300°F) is a charging/letdown mismatch. A charging/letdown mismatch involving all 3 pumps could allow 100 gpm injection. This rate would allow 10 minutes for operator action to preclude violation of the Appendix G curve in the event of a PORV failure to open. However, even without credit for operator action, the licensee's frequency estimate for an event that would challenge the vessel is about 1×10^{-5} per reactor year. This estimate assumes a PORV failure rate of about 10^{-1} per demand, a mismatch frequency of 10^{-2} per reactor year, and the fraction of time the plant would be operating in the temperature range per year (6 hours in 600 shutdown hours) or 10^{-2} per reactor year.

In view of the licensee's analysis and the historical data regarding challenges to systems with a pressurizer bubble and PORV (zero events), the staff considers that the screening value of 1×10^{-3} per reactor year discussed above for LTOP below 300°F is also conservative in the temperature range above 300°F.

Above 380°F and 450 psig the PORV is reset to 2500 psig. However in this range the vessel temperature is high enough that brittle fracture is of negligible concern.

III.3 LTOP Materials Evaluation

The licensee did not discuss materials aspects of LTOP events in their reports. The staff calculated the conditional probability of vessel fracture based on the peak pressure for the Yankee Rowe vessel using the methods set forth in Reference 11 and assuming RT_{NDT} is 320°F. An LTOP peak pressure in the range 1000-2000 psig has a conditional probability of vessel fracture in the range 10^{-3} to 10^{-2}.

III.4 LTOP Conclusion

Based upon a conservative screening value of $1x10^{-3}$ per reactor year for LTOP event frequency and a conditional vessel failure probability for LTOP events of 10^{-2} to 10^{-3}, the staff concludes that PTS events are bounding for brittle fracture considerations.

IV. UPPER-SHELF ENERGY EVALUATION

IV.1 Background

Reactor vessel beltline materials are required by Appendix G to 10 CFR Part 50 to have adequate fracture toughness. Specifically, beltline materials are required to have Charpy upper-shelf energy (USE) no less than 50 ft/lb throughout the life of the vessel. Otherwise, an analysis, approved by the staff, to demonstrate the existence of margins of safety against fracture equivalent to those of Appendix G of the ASME Code is required.

IV.2 Upper-Shelf Energy Events - Material Evaluation

In a letter dated May 1, 1990, the staff informed the licensee of the results of analyses that indicate that the USE for the Yankee Rowe vessel could be as low as 35.5 ft/lb. The staff specified the regulatory requirements that had to be met for vessels with USE below 50 ft/lb and provided the USE evaluation criteria based on current developments of the ASME Code. At present, these criteria have only been developed for ASME Code Service Levels A and B, e.g., Normal and Upset loading conditions. The staff believes that Service Level C and D, i.e., Emergency and Faulted conditions, criteria are unnecessary because, except for PTS and ATWS transients, Service Level C and D loads do not exceed level A and B loads. PTS events are discussed above. With regard to ATWS, the staff reviewed results of ATWS analyses which the licensee has submitted in 1974. The peak pressure estimated for a loss of feedwater ATWS was estimated to be 2820 psig. Since the licensee's Charpy USE analysis assu[illegible] an RCS pressure of 3437 psig the staff concludes that ATWS events are reasonably bounded by the licensees USE analyses.

The licensee performed an USE analysis for Normal and Upset loading conditions, i.e., ASME Code Service Levels A and B, using the ASME Code criteria now in preparation. The ASME code criteria now in preparation will require margins of safety against fracture equivalent to those required by the regulations. Based on a preliminary review of the licensee's analysis, it appears that the licensee's analysis satisfies the ASME code criteria for Service Levels A and B and provide margins of safety against fracture equivalent to thos[illegible] required by Appendix G of the ASME Code. The licensee also performed a low USE analysis for two of the PTS transients. The effects of low USE on crack arrest will also be considered in the PTS analysis being evaluated by the staff's contractor.

V. CONCLUSION

In order to address several NRC concerns with respect to th[illegible] requirement for reactor vessel fracture toughness for protection against [illegible]rized thermal shock events, the Yankee Rowe licensee has provided an [illegible] of the potential events leading to a challenge to the reactor [illegible] That analysis addressed both the probability of the initiating events as well as the probability of a pre-existing crack propagating through the vessel wall. The licensee also estimated the likelihood of challenges to the vessel from low temperature overpressurization events. As discussed above, there are a number of areas in which the staff concludes that additional safety margin or conservatism in the analysis would be appropriate; and that additional information to fully resolve the areas of concern is needed in order to determine whether longer term operation should be authorized. Actions required of the licensee during the next operating cycle are specified below. However, in the interim, the staff concludes that reasonable assurance of the public health and safety is provided since the potential for reactor vessel failure is very unlikely.

VI. FUTURE ACTIONS

In order for the licensee to demonstrate that longer term operation can be carried out without undue risk to the public health and safety, the licensee should provide the NRC, within 60 days after restart, a detailed plan of action. The following elements should be included in the plan:

VI.1 Short Term (Completed within 3 months)

1. Peer review of YAEC 1735, "Reactor Pressure Vessel Evaluation Report for Yankee Nuclear Power Station."

2. Revise fluence calculations.

VI.2 Long Term (Completed prior to Cycle 22 startup)

1. Develop inspection methods for the beltline welds and each beltline plate from the clad to 1 inch from the clad/steel interface to determine if the metal contains flaws.

2. Perform tests on typical Yankee Rowe base metal (0.18-0.20% Cu) to determine the effect of irradiation ($f = 1\text{-}5 \times 10^{19} n/cm^2$), austenitizing temperature (1650°F-1800°F) and nickel composition (0.18-0.70 percent) on embrittlement at 500°F and 550°F irradiation temperatures.

3. Determine composition of the circumferential weld metal in beltline by removing samples from the weld.

In addition, the licensee should install surveillance capsules in accelerated irradiation positions. The capsules are to include materials representing the beltline circumferential weld metal and upper and lower plates.

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TABLE I

LICENSEE AND STAFF ESTIMATES OF REFERENCE TEMPERATURE,

RT_{NDT} FOR THE YNPS BELTLINE MATERIALS IN 1990

YNPS Beltline Material	Unirradiated Ref. Temp. (°F)		Increase in Ref. Temp. Resulting from Irrad. (°F)		Ref. Temp., RT_{NDT} in 1990 (°F)	
	Staff Estimate	Licensee Estimate	Staff Estimate	Licensee Estimate	Staff Estimate	Licensee Estimate
Upper Plate	30	10	245	180	275	190
Lower Plate	30	10	325	173	355	183
Axial Welds	10	10	216	131	226	141
Circumferential Weld	10	10	320	219	330-370	229

TABLE 2

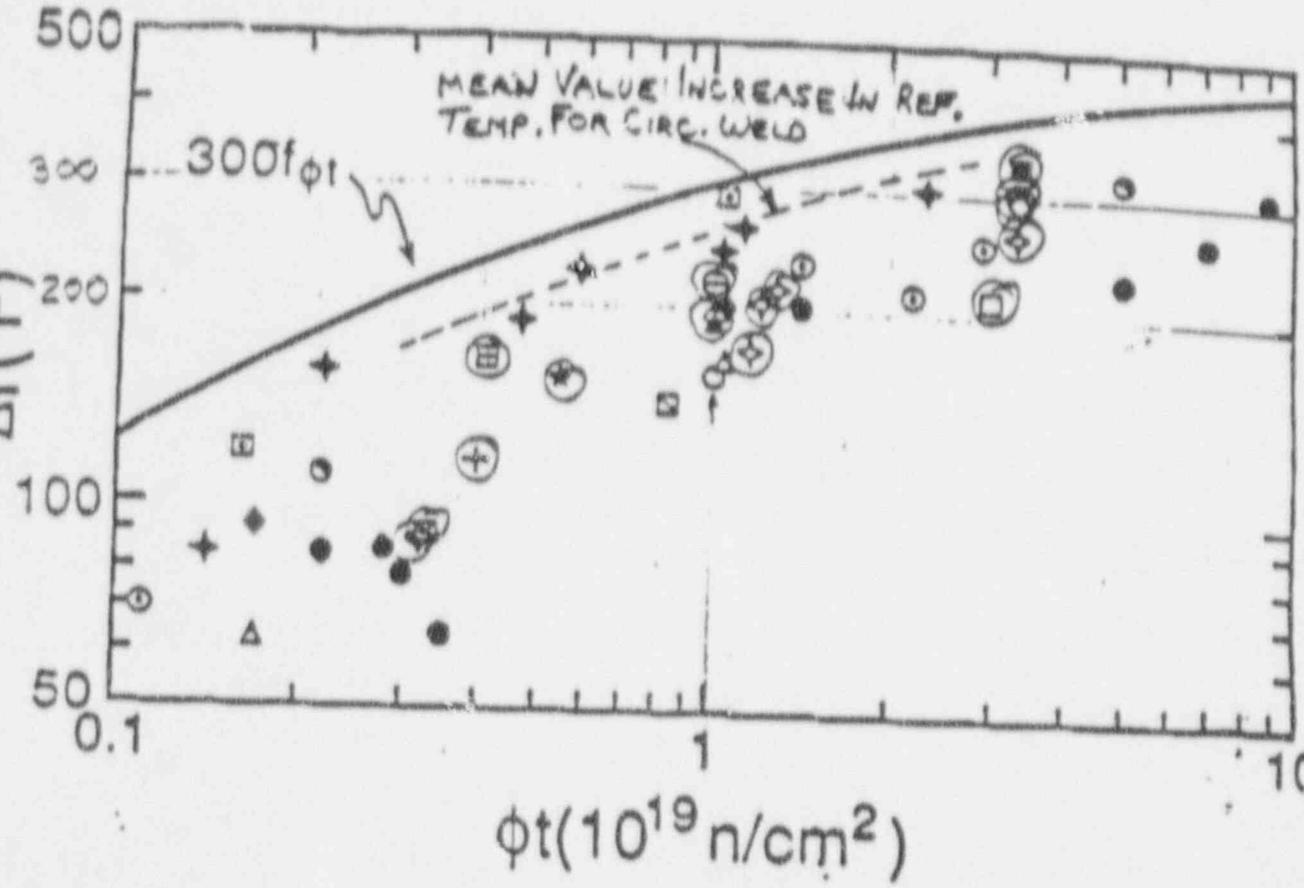
CHEMISTRY FACTOR FOR BASE METAL, °F

Copper, Wt-%	Nickel, Wt-% 0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	20	20	20	20	20	20	20
0.03	20	20	20	20	20	20	20
0.04	22	26	26	26	26	26	26
0.05	25	31	31	31	31	31	31
0.06	28	37	37	37	37	37	37
0.07	31	43	44	44	44	44	44
0.08	34	48	51	51	51	51	51
0.09	37	53	58	58	58	58	58
0.10	41	58	65	65	67	67	67
0.11	45	62	72	74	77	77	77
0.12	49	67	79	83	86	86	86
0.13	53	71	85	91	96	96	96
0.14	57	75	91	100	105	106	106
0.15	61	80	99	110	115	117	117
0.16	65	84	104	118	123	125	125
0.17	69	88	110	127	132	135	135
0.18	73	92	115	134	141	144	144
0.19	78	97	120	142	150	154	154
0.20	82	102	125	149	159	164	165
0.21	86	107	129	155	167	172	174
0.22	91	112	134	161	176	181	184
0.23	95	117	138	167	184	190	194
0.24	100	121	143	172	191	199	204
0.25	104	126	148	176	199	208	214
0.26	109	130	151	180	205	216	221
0.27	114	134	155	184	211	225	230
0.28	119	138	160	187	216	233	239
0.29	124	142	164	191	221	241	248
0.30	129	146	167	194	225	249	257
0.31	134	151	172	198	228	255	266
0.32	139	155	175	202	231	260	274
0.33	144	160	180	205	234	264	282
0.34	149	164	184	209	238	268	290
0.35	153	168	187	212	241	272	298
0.36	158	173	191	216	245	275	303
0.37	162	177	196	220	248	278	308
0.38	166	182	200	223	250	281	313
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

TABLE 3

COMPARISON OF REFERENCE TEMPERATURES ESTIMATED BY THE STAFF AND VALUES USED BY THE LICENSEE IN ITS SENSITIVITY STUDY

MATERIAL	REFERENCE TEMPERATURE ESTIMATED BY STAFF	MEAN VALUE REFERENCE TEMPERATURE USED IN LICENSEE"S SENSITIVITY STUDY
UPPER PLATE	275	280
LOWER PLATE	355	325
AXIAL WELD	226	222
CIRCUMFERENTIAL WELD	330-370	360



Cu/Ni/P	Prod. Form	Ref.	Cu/Ni/P	Prod. Form	Ref.
0.19/0.55/0.011	B	9	0.17/0.12/0.016	W	7
0.23/0.56/0.013	W	9	0.20/0.18/0.011	B	10
0.30/0.56/0.021	W	9	0.26/0.28/0.012	B	10
0.26/0.56/0.020	W	9	0.21/0.54/0.016	W	8
0.32/0.67/0.017	W	9	0.35/0.66/0.014	W	8
0.19/0.55/0.011	W	9	0.22/0.60/0.015	W	8
0.15/0.09/0.025	B	7	0.42/0.60/0.018	W	8
0.19/0.07/0.017	B	7	0.40/0.59/0.011	W	8
0.24/0.25/0.024	B	7	0.18/0.18/0.011 (YR-Plate)	B	1
0.21/0.17/0.033	B	7			

Figure 4 Shift data for 500 ± 10°F irradiations versus fluence and preliminary recommended trend curve.

FIGURE 1

Enclosure 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

June 25, 1991

Docket No. 50-029
(10 CFR Section 2.206)

Diane Curran, Esq.
Harmon, Curran, Gallagher & Spielberg
2001 S Street, N.W.
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Washington, D.C. 20009-1125

Dear Ms. Curran:

I am writing to acknowledge receipt of the "Petition for Emergency Enforcement Action and Request for Public Hearing" (Petition) submitted by you on behalf of the Union of Concerned Scientists and the New England Coalition on Nuclear Pollution (Petitioners). On June 4, 1991, the Petition was submitted directly to the Commissioners of the U. S. Nuclear Regulatory Commission (NRC). The Petition was filed in accordance with Section 2.206 of Title 10 of the Code of Federal Regulations (10 CFR 2.206) and thus should have been filed with the Executive Director for Operations. However, the Petitioners seek relief directly from the Commissioners because they believe that the NRC staff has failed to properly execute its responsibilities in this matter in permitting the Yankee Rowe Nuclear Power Station to continue to operate through Cycle 21 (approximately February 1992). The Petition has been referred to me for treatment under 10 CFR 2.206.

The Petition seeks the immediate shutdown of the Yankee Rowe facility of the Yankee Atomic Electric Company (licensee) based upon allegations that the Yankee Rowe facility is operating in violation of NRC requirements for reactor pressure vessel integrity and that the NRC staff's Safety Assessment of August 30, 1990, contains a number of deficiencies. The Petitioners argue these reasons prove that the continued operation of the Yankee Rowe facility poses a serious threat to public health and safety. The Petitioners further request that the Yankee Rowe facility remain shut down until it complies with regulatory requirements and that the Commission provide a public hearing, with rights of discovery and cross-examination, to determine the regulatory compliance before permitting the facility to resume operation.

The Petitioners allege specifically that the Yankee Rowe reactor pressure vessel failed to meet NRC requirements. The Petitioners argue that the Yankee Rowe facility does not comply with the requirements in 10 CFR 50.61 regarding reference temperature for reactor vessel material, the requirements in Appendix G to 10 CFR Part 50 regarding fracture toughness and the requirements in Appendix H to 10 CFR Part 50 regarding a surveillance program for reactor vessel material.

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On August 31, 1990, the NRC staff issued its "Safety Assessment of Yankee Rowe Vessel" (Safety Assessment) concluding that the Yankee Rowe facility could operate safely through Cycle 21. The Petitioners make specific allegations that the NRC staff's Safety Assessment is deficient. The Petitioners argue that the Safety Assessment contains errors and insufficient information in the assumptions underlying the calculations regarding the amount of neutron irradiation absorbed by the reactor vessel, the temperature of the metal during the time it is exposed to neutron irradiation and the chemical composition of the metal. In addition, the Petitioners argue that the Safety Assessment is inconsistent with the NRC policy on Safety Goals and that it failed to take into account the explicit recommendation of an NRC staff expert on reactor pressure vessel integrity that the Yankee Rowe facility not be permitted to operate.

The Petition presents no new information in regard to the integrity of the reactor vessel at the Yankee Rowe facility. The Petition expresses disagreement with the NRC staff's conclusions reached in the Safety Assessment that the Yankee Rowe facility was safe to operate through Cycle 21. The NRC staff has reviewed the Petition and has found no new information that would call into question the conclusions reached in its Safety Assessment. In making the Safety Assessment, the staff considered the views of NRC staff expert Dr. Randall as did the Advisory Committee on Reactor Safeguards (ACRS) which reported favorably regarding continued operation of the Yankee Rowe facility. (See letter of September 12, 1990, from ACRS, Enclosure 1.) The assertion that continued operation of Yankee Rowe constitutes a serious threat to the public health and safety because of the six alleged violations of NRC requirements is without merit for the following reasons.

The Petition indicates that the reference temperatures for the upper plate, the lower plate, and the circumferential weld exceed the screening criteria for pressurized thermal shock (PTS) in 10 CFR 50.61(b)(2). The licensee, as documented in Report YAEC No. 1735, July 1990, reports that the reference temperatures are below the PTS screening criterion. However, the NRC staff believes the PTS screening criterion may have been exceeded. That belief is based on conservatively considering the uncertainties associated with weld chemistry, irradiation temperature, grain size effects and flaw distribution as noted in the NRC staff Safety Assessment transmitted to the licensee by the letter of August 31, 1990. 10 CFC 50.61 does not require shutdown if the PTS screening criterion is exceeded. The NRC may, as specified in 10 CFC 50.61 (b)(5), on a case-by-case basis, approve operation of the facility at values of reference temperatures in excess of the PTS screening criterion. The rule requires the staff consider factors significantly affecting the potential for failure of the reactor vessel including the results of a probabilistic fracture mechanics analysis in reaching a decision to approve operation. The NRC staff also believes that the reference temperatures for axial welds in the upper and lower plates may also exceed the PTS screening criteria, as indicated in the October 9, 1990, memorandum to ACRS (Enclosure 2). The NRC staff requested the licensee to perform a probabilistic fracture mechanics analysis using conservative values of reference temperatures for PTS specified by the staff. Those reference temperatures exceed the screening criterion. The NRC staff reviewed the results from the probabilistic fracture mechanics analysis and considered the uncertainties resulting from low upper-shelf energy (USE) of the vessel materials, the lack of beltline inspection, and the reactor vessel's unique spot-welded cladding. Therefore, the NRC staff judged it to be prudent to assume the conditional probability of reactor pressure vessel failure to be in the range of 10E-1 to 10E-2 and the estimated frequency of the limiting PTS transient to be 10E-3 per reactor year. Based on this assessment the NRC staff authorized the licensee to operate the Yankee Rowe reactor vessel until the end of fuel Cycle 21.

The Petition indicates that the Yankee Rowe vessel upper plate is below the regulatory requirements for Charpy USE in 10 CFR Part 50, Appendix G, Section IV.A.1. However, Appendix G also indicates (1) that reactor vessels may be operated at lower values of Charpy USE, if operation is approved by the Director, Office of Nuclear Reactor Regulation, and (2) that lower values of Charpy USE provide margins of safety against fracture that are equivalent to those required by Appendix G of the ASME Code. The licensee provided a fracture mechanics analysis in Report YAEC No. 1735, July 1990, to demonstrate that the Yankee Rowe reactor vessel would have equivalent margins of fracture toughness to those required by Appendix G of the ASME Code with a Charpy USE of 35 foot-pounds. In its August 31, 1990, Safety Assessments, the NRC staff reviewed the licensee's analysis and approved the operation of the Yankee Rowe reactor vessel at levels of Charpy USE less than the limits in Section IV.A.1. of Appendix G to 10 CFR Part 50.

The Petition alleges that the NRC staff's Safety Assessment did not consider the revised neutron fluence estimates, the vessel operating temperature, and the vessel composition. The licensee revised its neutron fluence estimates in letters of September 28, 1990, and February 20, 1991. The NRC staff evaluated the neutron fluence estimates reported in the September 28, 1990, letter and documented its findings in the October 9, 1990, memorandum to ACRS (Enclosure 2). The peak neutron fluence estimates for the end of the current cycle (Cycle 21) reported in the licensee's February 20, 1991, letter are less than the values reported in its September 28, 1990, letter. Hence, the conclusions in the October 9, 1990, NRC staff memorandum apply to the end of the current fuel cycle. In preparing the Safety Assessment of August 31, 1990, the NRC staff considered the effect of the reactor operating temperature and the uncertainty in vessel composition that are discussed in the Petition. To account for the low operating temperature and the uncertainty in the vessel composition, the NRC staff increased the reference temperatures for the materials. These reference temperatures were evaluated as discussed herein.

The Petition alleges that the NRC staff did not consider that the beltline weld had not been inspected, had not received fracture toughness data from the licensee, and had not reviewed the licensee's analysis. The NRC staff was aware that the licensee has not volumetrically examined the beltline welds in the Yankee Rowe reactor vessel since the plant began operating. Therefore, the NRC staff in its probabilistic risk assessment assumed that flaws existed in the reactor vessel plates and beltline welds in order to account for the uncertainty resulting from the lack of volumetric examination. In Report YAEC No. 1735, July 1990, the licensee provided fracture toughness data and an analysis to demonstrate equivalent margins to Appendix G as noted herein. The NRC staff reviewed the data and analysis and provided its Safety Assessment in the letter of August 31, 1990.

The Petition asserts that the NRC staff's decision to allow the licensee to continue to operate Yankee Rowe is "flatly inconsistent with the Commission's 'Safety Goal' Policy that the risk of a severe accident should be kept to less than one chance in a million." The Petition indicates that this conclusion is based on the NRC staff's own calculation that the risk of pressure vessel rupture is between 5x10E-5 and 5x10E-6 and is thus greater than the Commission's large release guidance of 1x10E-6 per reactor year (that is, one in a million reactor years). The Safety Goal is not, and was never intended to be, a measure of adequate protection of public health and safety. Rather, the Safety Goal is a higher level of safety that the Commission believes the industry should strive to achieve. The Commission's Policy Statement on Safety Goals states the following:

> Current regulatory practices are believed to ensure that the basic statutory requirement, adequate protection of the public, is met.

The Policy statement further states the following:

> This statement of NRC safety policy expresses the Commission's views on the level of risks to public health and safety that the industry should strive for in its nuclear power plants.

The NRC staff's decisions regarding plant operation are based upon adequate protection of the public health and safety, not the Commission's Safety Goal Policy.

The Petition indicates that Yankee Rowe does not have a surveillance program as required by 10 CFR Part 50, Appendix H, and has not had its vessel ultrasonically inspected. The licensee discussed its surveillance and ultrasonic inspection program in Report YAEC No. 1735, July 1990. The licensee and the NRC staff used the data from the licensee's surveillance program to assess the integrity of the Yankee Rowe reactor vessel. The beltline welds in the Yankee Rowe reactor vessel were volumetrically examined by radiography as a part of its fabrication quality control. All flaws detected that exceeded the acceptance criteria were removed and repaired. Although the licensee has not ultrasonically examined the beltline welds since the plant has been in service, it has examined other similar welds and observed no unacceptable indications. However to account for the uncertainty that flaws might be present, the NRC staff, in its probabilistic risk assessment, assumed that flaws existed in the reactor vessel plates and beltline welds.

The Petitioners make the legal argument that compliance with NRC requirements is necessary to ensure that the Yankee Rowe facility operates safely. However, the failure to comply with a particular NRC requirement does not necessarily mean that there is no longer reasonable assurance of adequate protection of the public health and safety, particularly when the NRC staff has evaluated the area of alleged noncompliance and found that it does not pose an undue risk to the public health and safety. The NRC staff has evaluated the Yankee Rowe reactor vessel issues carefully and has concluded that the vessel condition continues to provide adequate protection of the public health and safety. In summary, the Petitioner's assertion that the alleged violations warrant immediate action to shut down Yankee Rowe is without merit.

Accordingly, Petitioners request for emergency relief is denied. As required by 10 CFR 2.206, the NRC will address the specific issues raised in the Petition within a reasonable time. Enclosure 3 is a copy of the Notice that is being filed with the Office of the Federal Register for publication.

Sincerely,

Thomas E Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Letter to K. M. Carr fm the ACRS, 10/9/90
2. Memo to Committee on Reactor Safeguards, 10/09/90
3. Related Federal Register Notice.

cc: Mr. George Papanic, Jr.
Dr. Andrew C. Kadak

Enclosure 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555

September 12, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: YANKEE ROWE REACTOR PRESSURE VESSEL INTEGRITY

During the 365th meeting of the Advisory Committee on Reactor Safeguards, September 6-7, 1990, we discussed the degree and consequences of the Yankee Rowe reactor pressure vessel embrittlement due to neutron irradiation. Our Subcommittee on Materials and Metallurgy discussed this matter with representatives of the NRC staff and the Yankee Atomic Electric Company during a meeting on September 5, 1990. We also had the benefit of the documents referenced.

It has recently come to the staff's attention that the reference temperature nil ductility transition (RT_{NDT}) of parts of the Yankee Rowe pressure vessel may substantially exceed the temperature limits for action delineated in the pressurized thermal shock (PTS) rule (10 CFR 50.61). The main reason is that the Yankee Rowe core inlet temperature is about 50°F lower than that of other plants. Another reason is the higher nickel content of the lower vessel plate. These increase the rate of rise in RT_{NDT} with fast neutron irradiation.

The exact value of RT_{NDT} for the vessel is uncertain because of:

- Uncertainty in the copper and nickel content of the circumferential weld near the reactor vessel beltline.
- The absence of surveillance data for areas that appear to have the largest shift in RT_{NDT}, namely the circumferential weld and the lower plate of the vessel.

Assurance of vessel integrity is further hindered by:

- The absence of any inservice inspection for flaws in the reactor vessel beltline region. Such inspection has been infeasible due to the design of the vessel internals.
- Relatively low toughness (low upper shelf energy) of the plate and welds near the core.

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Analysis of the various safety issues involved leads to the conclusion that PTS is the issue of most concern. One bright spot in this picture is that several features of the plant's design make it less susceptible to overcooling events than more modern plants.

The licensee and the staff have both arrived at estimates of the shift in RT_{NDT}. Both agree that the circumferential weld and the lower plate of the pressure vessel have the highest RT_{NDT}. However, in each case their estimates differ by about 150°F. The licensee's representatives argue that due to the particular microstructure of the steel in the vessel, the shift in RT_{NDT} is independent of irradiation temperature and nickel content. We do not believe these arguments are valid, and agree with the staff that temperature and nickel effects must be included in a valid estimate of the shift in RT_{NDT}. An additional difference between the staff and the licensee concerns estimates of the copper content of the circumferential weld. There being no measurements for the composition of the circumferential weld and a large spread in copper values found in other plants, the staff prefers to choose a bounding value. The applicant chose more of an average value. In view of the uncertainty in the value for the Yankee Rowe vessel, we would choose the staff's bounding value.

Given that RT_{NDT} values for parts of the vessel probably exceed those requiring action under the PTS rule, is there significant risk in operating the plant? The low probability of a PTS challenge leads to a low risk, even with a high RT_{NDT}. Thus, we agree with the staff that operation for one more cycle is acceptable, provided the licensee initiate an active program to better characterize the material in the vessel near the reactor vessel beltline. To do this the staff requires determination of the composition of the circumferential weld metal in the beltline by removing samples from the weld and development of an inspection method for the beltline welds and plate to depths of an inch below the inside surface of the vessel. Both of these have been required by the staff for completion before the startup of the 22nd fuel cycle (now scheduled to begin in early 1992). It is not clear that both can be achieved in that time, but certainly they should be accomplished in two fuel cycles.

The staff also requires "tests on typical Yankee Rowe base metal" to determine the effect of irradiation, austenitizing temperature and nickel content on embrittlement. It is doubtful that any tests that the licensee could perform during the next fuel cycle would convince us that the effects of temperature and nickel on embrittlement are substantially different from those established by the much more extensive studies already available. The effects are not well understood, and we believe prudence dictates tending more toward bounding values rather than best estimates based on limited new data that may become available.

However, the above will not adequately address the long-term operation of the plant. This is the lead PWR plant in the industry's Plant Life Extension (PLEX) program, and long-term operation with such large uncertainties in vessel integrity is unacceptable. The extended operation of this plant would be acceptable only if:

- A state-of-the-art ultrasonic inspection can be done on essentially all of the radiation affected inner surface of reactor pressure vessel, e.g., one that complies with Appendices VII and VIII of Section XI of the ASME Code. This inspection should also check for significant thinning in the lower head as a result of loose parts (irradiation capsules). Continued operation would be dependent on the absence of significant flaws.

- A reanalysis of the PTS question is made using well established compositions for the material in the beltline region, or using limiting values of copper and nickel. This analysis should also include the fact that the crack arresting ability of such material will be lower than more modern steel because of its low upper shelf energy. Such an analysis must show acceptable risk.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:

1. Letter dated July 5, 1990 from John D. Haseltine, Yankee Atomic Electric Company, to Richard Wessman, NRR, transmitting Reactor Pressure Vessel Evaluation, dated July 9, 1990

2. Letter dated August 31, 1990 from Thomas E. Murley, NRR, to Andrew C. Kadak, Yankee Atomic Electric Company, Subject: Yankee Rowe Reactor Vessel, with Enclosure